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DETERMINATION OF LEU CORE SAFETY RELATED PARAMETERS OF NIRR-1 USING WIMS AND CITATION CODES

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ABSTRACT
The Nigeria Research Reactor-1 (NIRR-1) is a Chinese Miniature Neutron Source Reactor (MNSR) with a nominal thermal power of 31.1 kW which is equivalent to thermal neutron flux of 1 x 10^{12} nv. Its core conversion from HEU (~90%) to LEU (~20%) has been considered. The material composition required for this conversion has been investigated, maintaining as far as possible, the technical specifications and operating procedures of the present HEU core. A simplified scheme of WIMSD and CITATION was employed to investigate the Safety margin parameters like effective multiplication factor (k_{eff}), excess reactivity (ρ_{ex}), shutdown margin (SDM) and Safety Reactivity Factor (SRF) at LEU since these parameters can change with changes in the material composition of the reactor core. Two fuel types namely the present UAl4 and a potential UO_2 fuels were considered. The LEU results obtained here show ρ_{ex} of ~3.5mk for both fuel types, SDM of 2.93mk for UO_2 and 2.18mk for UAl_4 at 20% enrichment. SRF values of about 1.8mk for UO_2 and about 1.6mk for UAl_4, which compares favorably with the present UAl_4 fuel at HEU. The results are quite remarkable and have shown the potentials of the developed scheme in core conversion studies.

KEYWORDS: Performance Test; Steady-State Conditions; Neutronics Parameters

INTRODUCTION

The international Reduced Enrichment for Research and Test Reactors (RERTR) program has recently advanced a cause for conversion of reactors with highly enriched uranium (HEU) to low enriched uranium (LEU); Azande, *et al*. (2010). Efforts are underway, at the Center for Energy Research and Training (CERT) Zaria, to convert the Nigeria research reactor-1(NIRR-1); (Azande, *et al*. 2010, Jonah, *et al.*, 2009; Balogun, 2003) from HEU to LEU in conformity with this global trend. The parameters investigated here are critical to the operational safety of the reactor and may change with reduced enrichment. The parameters were investigated for LEU (< = 20%) maintaining the original structural design specifications of the reactor.

NIRR-1 assembly is composed of a reactor core, beryllium reflectors and a centrally located control rod made of 3.9mm diameter and 266mm long cadmium having a stainless steel cladding of 0.5 mm thickness reactivity equivalence of about 7mk. The core is an under-moderated array with hydrogen to U-235 atomic ratio of about 197. The excess reactivity of cold-clean-core is about 3.77mk. The core consists of 344 fuel pins (Jonah, *et al.*, 2009) arranged in concentric arrays and forms a square cylinder with a diameter of 231mm and a height of 230mm. The total fuel loading in the fresh core is about 1kg of U-235. The fuel pins are secured by upper and lower grid plates to form a fuel cage. The fuel pins are fixed to the lower grid plate by slightly conical self-locking fittings and are free to expand through the upper grid plate. Four out of the 355 lattices provided in the grid plates are used for fixing stainless steel tie rods to keep the fuel cage intact. Six other lattices are occupied by aluminium dummy pins. The central lattice is used for the central control rod guide tube, which is provided to facilitate the movement of the control rod. A full description of NIRR-1 is published elsewhere by Jijin (1993). Figure: 1 shows the core configuration.

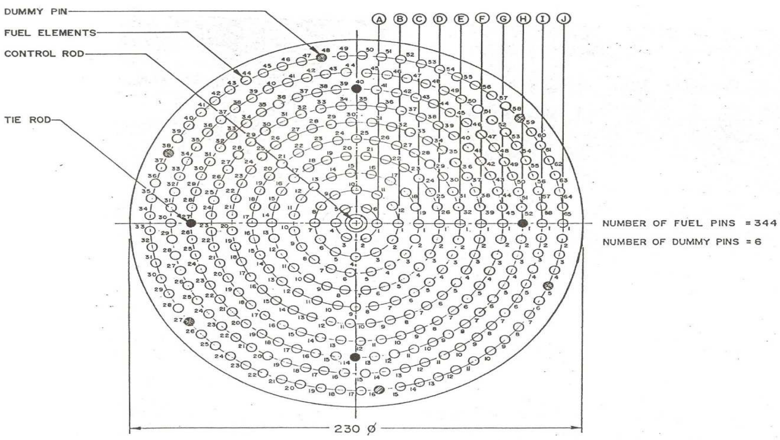

Figure 1: NIRR-1 Core Configuration

A weight of about 850g is attached to the control rod to facilitate its movement. The control rod moves through a hole in the top reflector shim tray having a travel length of about 230mm. Loss of power to its drive mechanism releases the rod to fall into the core under gravity; the total rod drop time from fully withdrawn position to fully inserted position is 26.5s. The reactor is fueled with HEU fuel pins which are about 90.2% enriched in U-235. The fuel meat is uranium-aluminium alloy (UAl_4). The fuel pins are 4.3 mm in diameter with an active length of 230mm with a cladding of 0.6mm thick aluminium alloy (303-1 Al). Each fuel pin contains 3.169g of uranium with about 2.9g of U-235 and the surface is finished with anodic treatment of 15µm thickness; (Jijin, 1993, SAR, 2005).

THEORITCAL CONSIDERATION

Neutron leakage was reduced by surrounding the reactor core with additional moderating material (beryllium) which acts as a reflector, returning some of the escaping neutrons to the reactor assembly. By introducing the infinite multiplication factor, k_∞ for the extension, and therefore without neutron leakage, one can express the k-values in terms of the effective macroscopic cross sections of the system.

$$k_\infty = \frac{production}{absorption} = \frac{v(\Sigma_f)}{(\Sigma_a)} \qquad 1$$

And

$$k_{eff} = \frac{production}{absorption+leakage} = P_{nl} k_\infty \qquad 2$$

Where v is the average number of neutrons, $\Sigma_f$is macroscopic fission cross section, Σ_a is the total absorption cross section. These expressions for k_∞ and k_{eff} appear deceptively simple as the fundamental quantities, that is the effective macroscopic cross sections of the entire system and the non-leakage probability P_{nl}, are of course unknown; and their determination would be equivalent to the solution of the fundamental transport equations. Diffusion theory was used to derive formulae for k_∞ and k_{eff} that are valid under certain conditions and for specified simple geometries and most generally, k_{eff} can be written in an integral form; (Cullen *et al.,* 2003) as:

$$k_{eff} = \int_v \int_0^\infty \int_E \int_\Omega v \Sigma_f \phi d\Omega dE dt dv \cdot \left[\int_v \int_0^\infty \int_E \int_\Omega \Sigma_a \phi + (\Delta.\vec{j}) d\Omega dE dt dv\right]^{-1} \qquad 3$$

Where ϕ is the angular neutron flux, with energy E moving in the direction Ω.

The resonance region in WIMS is defined to lay between 4.0eV and 183.2KeV. Resonance absorption above 183.2KeV is regarded as unshielded and infinite dilution cross sections are used above this energy for all resonance absorbers. The methods used in WIMS for regular lattices of rods or finite plates, clusters of rods or bundles of plates are similar in that equivalence theorems, (Newton, 2004) are used which relate the

heterogeneous problem to an equivalent homogeneous problem in each case. The equivalence theorems take account of non-hydrogenous moderators which may be mixed with the fuel such as oxygen or carbon, and also include a first order correction for the interaction associated with the presence of several resonant isotopes. The library of homogeneous resonance integrals has been formed through the use of the Spin Dependent Recombination (SDR) program (Brissenden and Durston, 1965).

To interpret which value of potential scattering is appropriate to a specific problem, WIMS uses sub-group theory; (Newton and Hutton, 2002) in which the heterogeneous problem is reduced to a homogeneous problem. The SDR program solves the Chernick and Vernon integral equations in the form

$$I(\sigma_p) = \int \sigma_a(u)\varphi(u)du = \frac{1}{\delta u}\int \frac{\sigma_p\sigma_a(u)}{\sigma_p + \sigma_a(u)}du = \sum_j \frac{\omega_j\sigma_p{\sigma_a}^j}{\sigma_p + {\sigma_a}^j} \qquad 4$$

Where

$U = lethergy$

$\sigma_a = absorption\ cross - sections$

$\sigma_p = potential\ scaterring\ cross - sections$

$\omega_j = sub - group\ weight$

${\sigma_a}^j = sub - group\ absorption\ cross - sections$

METHODOLOGY

The Nigeria Research Reactor-1 (NIRR-1), which is currently in its first fuel cycle, was licensed to operate at 31.1kW; (Balogun and Jonah, 2004). It is the 8th commercial miniature neutron source reactor (MNSR) designed by China Institute of Atomic Energy (CIAE). First criticality was achieved on 03 February 2004 and has been operated safely; (CERT, 2004).

To model the neutronics of NIRR-1, a simplified scheme of WIMS and CITATION (Azande *et al.*, 2010; Balogun, 2003) system of computer codes were employed. WIMSD (Halsall, 1980; Askew *et al.*, 1966) uses 69 group multi region integral transport theory to solve the neutron transport equation for the lattice cells. The group collapsed cross sections were obtained from WIMSD and were used as input for CITATION (Fowler *et al.*, 1981), which uses finite difference scheme to solve the neutron diffusion equation from one up to three dimensions.

In order to generate the group constants for fuel region, control rod, control rod follower, reflectors and other non-fueled regions, unit cell calculations based on Wigner-Seitz (Lamarsh, 1966) cell modeling was used. The lattice structure of the core was represented in the form of a super cell, (Iqbal *et al.*, 2007) which comprises of a fuel, cladding, moderator, dummy and a structure zone. The 69 group WIMSD library was collapsed to produce a four group self shielded cross-section data set. Based on energy limit for neutron up scattering, unresolved resonances and inelastic scattering, the energy group structure with boundaries as shown in Table 1 were chosen. Group constants generated by lattice cell calculations were used to represent different core regions in CITATION. Macroscopic cross-sections were generated using WIMSD while CITATION was used for core modeling.

Table 1: *Energy group structure used in WIMSD for condensation in generation of group constant*

Group number	Energy (eV)	
	Upper Limit	Lower Limit
1	1×10^7	1.353×10^6
2	0.821×10^6	9.118×10^3
3	5.530×10^3	0.780
4	0.625	0.180

RESULTS AND DISCUSSION

The excess reactivity of UAl_4 was found to be 3.57mk, 3.52mk, and 3.86mk at 10%, 15% and 20 % enrichments respectively. UO_2 gave 3.53mk at 10%, and 3.55mk for both 15% and 20%. This is in good agreement with the design specification of 3.5mk – 4.0mk for MNSR as reported by (Chengzen, 1993), the Final Safety Analysis Report (SAR, 2005) and HEU values reported by (Balogun, 2003). The results from Tables 1 and 2 show that the SDM values are within the range of 2.0 and 3.5 mk which is in good agreement with the SDM specification for the MNSR.

Table 2: Results for uranium-aluminium (UAl_4) fuel at Low enriched uranium

Enrichment (%)	k_{eff} Value	Core excess reactivity (mk)	Safety Reactivity Factor (SRF)	Shut Down Margin (SDM) (mk)
10	1.0035	3.57	1.59	2.08
15	1.0035	3.53	1.70	2.46
20	1.0039	3.86	1.60	2.18

Table 3: Results for uranium-oxide (UO_2) fuel at Low enriched uranium

Enrichment (%)	k_{eff} Value	Core excess reactivity (mk)	Safety Reactivity Factor (SRF)	Shut Down Margin (SDM) (mk)
10	1.0035	3.53	1.67	2.35
15	1.0036	3.55	1.77	2.73
20	1.0036	3.55	1.83	2.93

On safety criteria, the Safety Reactivity Factor (SRF) of NIRR-1 should be greater than 1.5 as reported in Balogun (2003). This implies that the greater the ratio of the control rod worth to the core excess reactivity is, the better and safer the reactor would be. This would give room for a wider shutdown margin enabling the reactor to be gradually shutdown when in operation. The results obtained for UAl_4 and UO_2 at LEU compares favorably with HEU values given in Balogun (2003). The variation of SRF with enrichment for UO_2 (Fig 1) indicates that SRF increases steadily with increase in enrichment while that of UAl_4 (graph 2) did not show this characteristic rather oscillates between 1.5 and 2.0. This behavior of UAl_4 is due to the compact nature of the core, which was designed to cause insufficient thermal circulation of coolant in the core; (Ahmed *et al.*, 2008). However, both at LEU and HEU, Fig 1 and 2 are within 1.5 and 2.0.

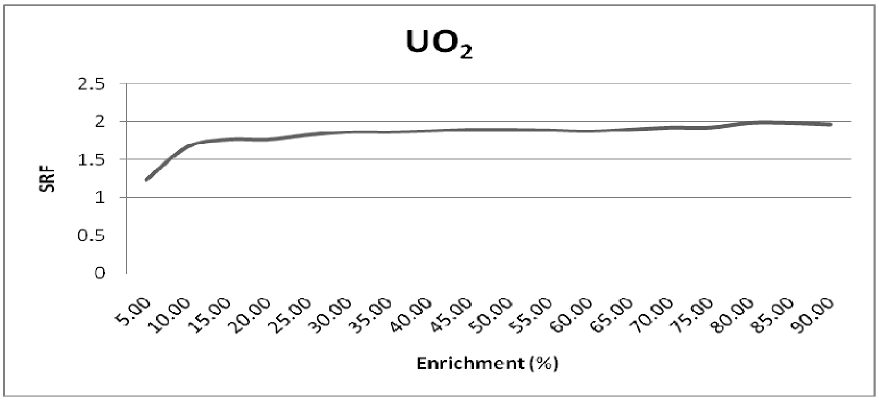

Fig 1: A plot of safety reactivity factor against enrichment for UO_2

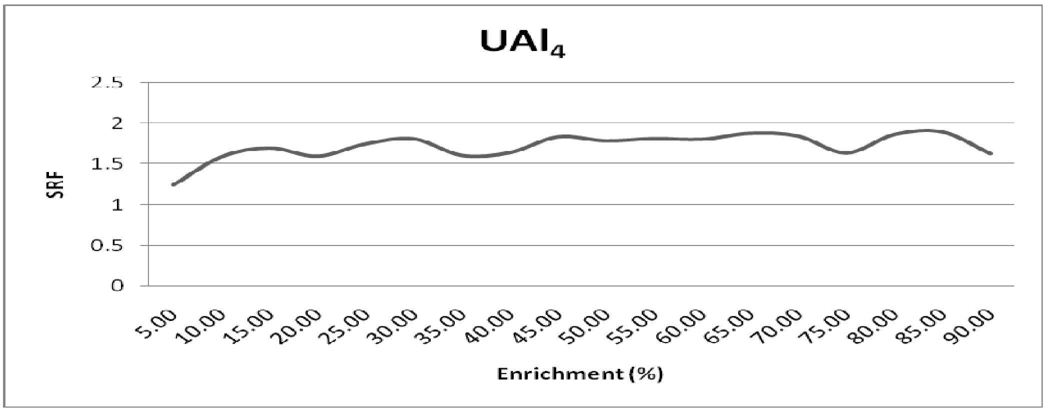

Fig 2: A plot of safety reactivity factor against enrichment for UAl_4

The safety related parameters of NIRR-1 at LEU core were investigated in this work using a simplified scheme of WIMS and CITATION codes. The results obtained show that the SRF of 1.59 mk at 10%, 1.70mk at 15% and 1.60 mk at 20% for UAl_4 and 1.67mk at 10%, 1,77mk at 15% and 1.83mk at 20% enrichment for UO_2 were obtained. The shutdown margin (SDM) for the respective enrichments are 2.08mk, 2.46mk and 2.18mk for UAl_4 and 2.35mk, 2.73mk and 2.93mk for UO_2, with core excess reactivity values in good agreement with the Chinese design specifications for MNSR (Jijin, 1993). The investigation indicates that NIRR-1 satisfies the design limits at LEU with increasing core loading of U-235 for NIRR-1.

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